NON-PUBLIC?: N
ACCESSION #: 9411210290
LICENSEE EVENT REPORT (LER)

FACILITY NAME: LaSalle County Station Unit 1 PAGE: 1 OF 3

DOCKET NUMBER: 05000373

TITLE: Scram Due to Reactor Water Level Control Signal Loss to the 1B Turbine Driven Reactor Feed Pump
EVENT DATE: 07/05/94 LER #: 94-010-01 REPORT DATE: 11/02/94

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 056

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Jack Otlewis, System Engineer, TELEPHONE: (815) 357-6761 Extension 2447

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SJ COMPONENT: CBD MANUFACTURER: B040

REPORTABLE NPRDS: Yes

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At 0336 hours on July 5, 1994, Unit 1 scrammed from a power level of 601 MWe and 56.4% Core Thermal Power. The unit had been in Operational Condition 1 (Run) with the 1B Turbine Driven Reactor Feed Pump (TDRFP, FW)SJ! in 3 element control and the Motor Driven Reactor Feed Pump in manual. The 1A TDRFP was unavailable pending additional post modification testing of the feedwater control system. Just prior to the scram, a loss of the Reactor Water Level Control signal caused the 1B TDRFP to ramp to minimum speed, resulting in a decrease in Reactor Vessel level. The reactor scrammed on a Reactor Low Water Level Scram (12.5 inches). The lowest reactor vessel level during the transient was -6 inches. Normal level was restored using the Motor Driven Reactor Feed Pump. Subsequent investigation determined the cause of the feedwater transient was due to a failure of the Manual Control Board Unit.

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) as an event that resulted in automatic actuation of the Reactor Protection System. This revision updates the Corrective Action taken with respect to the failure of the Manual Control Board Unit.

END OF ABSTRACT

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as XX!.

A. CONDITION PRIOR TO EVENT

Unit(s): 1 Event Date: 7/05/94 Event Time: 0336
Hours
Reactor Mode(s): 1 Modes(s) Name: Run Power Level(s):
56%

B. DESCRIPTION OF EVENT

Operators were pulling rods to increase power after Reactor Recirculation (RR)AD! pump upshift. At 0336 hours with Unit 1 at 601 MWE and 56.4% Core Thermal Power the 1B Turbine Driven Reactor Feed Pump (TDRFP, FW)SJ!, while in 3 element control (Motor Driven Reactor Feed Pump in manual) lost a Reactor Water Level Control signal causing the 1B TDRFP to ramp to minimum flow. The Operator attempted, but was unable, to gain control of Reactor Vessel water level. The Reactor scrammed on 12.5 inch Low Reactor Water Level. Procedure LGP-3-2, "REACTOR SCRAM", was entered. The lowest Reactor Pressure Vessel (RPV) level was -6 inches. Procedure LGA-01, "RPV CONTROL" was entered and level was restored with the Motor Driven Reactor Feed Pump. The "A" Electro Hydraulic Control (EHC, EH)TG! Pump also tripped following the 151 switchgear fast transfer.

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) as an event that resulted in automatic actuation of the Reactor Protection System.

C. APPARENT CAUSE OF EVENT

The cause of the Reactor Low Water Level Scram was loss of feed flow to the Reactor Pressure Vessel (RPV) following the loss of control signal to the 1B TDRFP. The loss of this control signal resulted in ramping the 1B TDRFP output flow to zero and subsequently the Reactor Low Water Level Scram. Manual control of the 1B TDRFP did not respond due to the failed component. The cause of the 1B TDRFP control signal failure has been determined to be a Manual Control Board Unit (1C34-K653B) which provides the output signal from the Reactor Water Level Control (RWLC, LC)BD! System to the 1B TDRFP Love joy Speed Control System. Both automatic and manual signals are processed through this card.

D. SAFETY ANALYSIS OF EVENT

The safety consequences of this event were minimal. All Engineered Safety Feature (ESF) actuations occurred as designed on the Reactor Low Water Level Scram. Loss of total feedwater flow with the reactor at 104.8% power, analyzed as a moderate frequency transient in Section 15.2.7 of the Updated Final Safety Analysis Report (UFSAR), bounds this event.

E. CORRECTIVE ACTIONS

The 1C34-K653B circuit card was replaced. The failed circuit card was sent to the manufacturer, Bailey Controls Company, to determine the failure cause and to repair the card. Troubleshooting revealed a cold solder connection on one of the board components. The connection was repaired and the board was returned to the station.

The 1A EHC Pump trip was found to be due to switchgear 131A breaker trip device having two leaking dash pots. These were repaired. Trip devices of this type used in safety related applications had previously been replaced with a more reliable device. The non-safety trip devices are currently undergoing evaluation for replacement with a more reliable device.

F. PREVIOUS EVENTS

The following scrams have occurred due to feedwater control:

LER No. Title

373/91-006 Reactor Scram on Low Reactor Vessel Water Level Due to Loss of "A" Turbine Driven Reactor Feedwater Pump

Caused by Control Valve Closure

373/93-011 Unit 1 - Manual Scram Due to Disconnected Linkage on Valve Positioner on a Heater Drain Valve

G. COMPONENT FAILURE DATA

Manufacturer Nomenclature Model Number MFG Part Number

Bailey Manual Control Board 722 722001AAAA1 Unit

ATTACHMENT TO 9411210290 PAGE 1 OF 1

Commonwealth Edison
LaSalle County Nuclear Station
2601 N. 21st Road
Marseilles, Illinois 61341
Telephone 815/357-6761

November 2, 1994

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Licensee Event Report #94-010-01, Docket #050-373 is being submitted to your office in accordance with 10CFR50.73(a)(2)(iv). This report has been updated to document results of testing performed by the vendor on the failed Manual Control Board Unit Circuit Board.

D. J. Ray

Station Manager
LaSalle County Station

DJR/lja

Enclosure

cc: NRC Region III Administrator
NRC Senior Resident Inspector
INPO - Records Center
IDNS Resident Inspector
IDNS Senior Reactor Analyst

Nuclear Licensing Administrator

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